Entergy

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J. E. Pollock
Site Vice President

NL-09-072

July 6, 2009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Licensee Event Report # 2009-003-00, "Manual Reactor Trip Due to Steam Generator 33 High Water Level Caused by a Failed 33 Main Feedwater Regulating Valve"
Indian Point Unit No. 3
Docket No. 50-286
DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2009-003-00. The attached LER identifies an event where the reactor was manually tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A) . As a result of the reactor trip, the Auxiliary Feedwater system was actuated which is also reportable under 10 CFR 50.73(a)(2)(iv)(A). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP3-2009-02368.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Licensing at (914) 734-6710.

Sincerely,

D. Mayer for

JEP/cbr

cc: Mr. Samuel J Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 3
Mr. Paul Eddy, New York State Public Service Commission
LEREvents@inpo.org

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NRR

NRC FORM 366 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 8/31/2010

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Manual Reactor Trip Due to Steam Generator 33 High Water Level Caused by a Failed 33 Main Feedwater Regulating Valve

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
5	15	2009	2009-	003 -	00	7	06	2009		05000
									FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Tom Foley, System Engineer	(914) 734-6760

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
D	SJ	V	B455	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

16. ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines)*

On May 15, 2009, Control Room operators initiated a manual reactor trip for high steam generator (SG) 33 water level which could not be corrected due to a failed open 33 main feedwater (FW) regulating valve (FCV-437). All control rods fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser. There was no radiation release. The Emergency Diesel Generators did not start as offsite power remained available. In response to the event main FW isolated and the Auxiliary Feedwater System automatically started. The direct cause of the high SG level was the positioner feedback linkage on FCV-437 became disconnected causing the valve to open. The cause of the positioner feedback link disconnection was inadequate procedure direction and written instructions. The valve maintenance procedure did not contain steps to perform wrench tightness of the rod jam nuts after positioner linkage was disconnected, and then reconnected during valve overhaul. There were also no steps in the diagnostic test procedure that directed workers to verify wrench tightness of the jam nuts. A contributing cause was ineffective use of Human Performance (HP) tools. Corrective actions included repair of the valve positioner linkage and verification of other position linkage assemblies. The valve overhaul procedure and diagnostics procedure will be revised to ensure direction to wrench tighten jam nuts and verify installation of lock washers on jam nuts, wrench tighten connection of linkage to the valve yoke and positioner, wrench tighten connection of the turnbuckle to yoke and verify minimum thread engagement of the rod to turnbuckle. The event had no effect on public health and safety.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

On May 15, 2009, while at approximately 100% steady state reactor power, Control Room operators received a steam generator (SG) {AB} level control deviation alarm for rising SG 33 water levels with a 33 main feedwater (FW) {SJ} regulating valve (33 FRV) not responding while in automatic. Control room operators placed the 33 FRV in manual but were unable to control the rising 33 SG water level. When the SG water level approached the SG high level set point, Control Room Operators initiated a manual reactor trip (RT) {JC} at approximately 01:53 hours, in response to the rising SG water level which could not be corrected due to a failed open 33 FRV (FCV-437) {FCV}. All control rods {AB} fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser {SG}. There was no radiation release. The Emergency Diesel Generators {EK} did not start as offsite power remained available. In response to the event, main FW isolated and the Auxiliary Feedwater System {BA} automatically started. The event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP3-2009-02368. A post transient evaluation was initiated and completed on May 15, 2009.

Investigation into the event discovered that the 33 FRV (FCV-437) positioner feedback link had disconnected. The linkage was observed to be disconnected on the bottom where the connecting link (threaded rod) attaches to the lower turnbuckle with the jam nut still on the rod with a star lock washer still in place. The upper connection, where the connecting link attaches to the upper turnbuckle, was connected to the turnbuckle and did not have a locking star type washer installed. The top jam nut was run down the shaft approximately two inches from the turnbuckle.

Main FW regulating valve (FRV-33) BFD-FCV-437 is an air operated flow control globe valve (AOV) manufactured by Copes Vulcan {C635}, Model D-100-160 actuator and valve. The valve fails closed on a loss of air and has a Bailey Model AV-1 positioner (ABB Brown Boveri) {B455}. The valve's safety function is to close to terminate FW flow to the SG. The valve will close by venting air pressure on receipt of a SG High Level signal, a Safety Injection signal or a RT signal. The elements that encompass the control loop are the Controller, current to pneumatic (I/P) Converter, Positioner, and Actuator/valve. The controller provides a signal to the I/P based on steam flow, FW flow, and SG level offset. The I/P converts the controller signal to a pressure which modulates the positioner to move the valve to the demanded position. The type AV positioner operates by balancing opposing forces. A balance beam hinged at one end and connected to the pilot valve at the other end is acted upon by the upward force of the signal diaphragm assembly and the downward force of the range spring. The range spring is a function of the shape and position of the cam. The cam is coupled through the cam shaft that is connected through a linkage assembly. Varying the input signal changes the force exerted by the signal diaphragm, moving the balance beam, in turn moving the pilot valve. The pilot valve supplies and/or exhausts air to the actuator that changes its position.

The loss of the feedback linkage resulted in the opening of valve BFD-FCV-437 and a subsequent decrease in control valve signal. Although the control signal demanded the valve to close, the positioner continued to send air to open the valve based on the feedback mechanism being in the closed position. Consequently, automatic and manual control of BFD-FCV-437 was lost and the valve continued to open allowing continued FW flow to the SG causing its level to rise.

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A review of the work history for the FRVs determined FCV-437 and FCV-417 were worked on in a recently completed cycle 15 refueling outage which ended April 15, 2009. Valve work was performed by a vendor and included an overhaul per procedure 0-VLV-416-AOV and valve diagnostics per procedure 0-VLV-404-AOV. A six year overhaul and a two year diagnostic testing was performed on valve FCV-437 and completed on April 4, 2009. A review of the work package for the valve overhaul and diagnostics did not identify any steps that directed wrench tightening of the jam nuts on the threaded rod or a description of the jam nuts or lock washers. The diagnostics procedure contained a step on an inspection checklist that mentioned the jam nuts. There were no steps to verify installation of a star lock washer or locking device in the overhaul or diagnostic procedures. When performing valve overhauls the linkage and positioner are removed and the jam nuts can be disturbed where they could be unknowingly loosened. When the linkage and positioner are reinstalled after completing the valve work, failure to check and retighten the jam nuts can result in the nuts loosening due to vibrations during operation. The FRVs have constant vibration where the linkage is installed and if the jam nuts are not adequately wrench tightened they could vibrate loose and cause the threaded rod to unscrew and come out of the turnbuckle and disconnect.

An extent of condition (EOC) review was performed and verified the remaining FRVs (FCV-417, FCV-427, FCV-447) had star washers, there was thread engagement within the linkage assembly, and the jam nuts were tight. The linkage assembly for FCV-417 was disassembled and re-assembled as part of this review as this valve was overhauled by the same vendor during the cycle 15 refueling outage. The corresponding Unit 2 FRVs were also inspected and verified to be operating properly with no signs of linkage unthreading.

Cause of Event

The direct cause of the failed BFD-FCV-437 was a disconnected positioner feedback link. The root cause of the positioner feedback link disconnection was inadequate procedure direction and written instructions. Maintenance procedure 0-VLV-416-AOV (FRV Inspection and Overhaul) did not contain steps to perform wrench tightness of the rod jam nuts after positioner linkage was disconnected/reconnected during valve overhaul. There were also no steps in the diagnostic test procedure 0-VLV-404-AOV (AOV Diagnostics) that directed workers to verify wrench tightness of the jam nuts. A contributing cause was ineffective use of Human Performance (HP) tools. Valve technicians failed to utilize HP tools such as questioning attitude and self checking to determine what the intent of the procedure 0-VLV-404-AOV Attachment 5 checklist linkage inspection was and ensure the rod jam nuts were wrench tight.

Corrective Actions

The following corrective actions have been or will be performed under Entergy's Corrective Action Program to address the cause and prevent recurrence:

- The valve positioner linkage was repaired on BFD-FCV-437 and valve BFD-FCV-417 was disassembled, linkages checked and verified correct, re-assembled and diagnostic testing performed.
- Critical valves worked by the same contractor team were inspected and there were no instances of loose linkages or jam nuts.
- The total AOV work scope for refueling outage 3R15 was reviewed and critical valves inspected. The inspection results did not indicate any instances of loose linkages or jam nuts.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

- Maintenance and I&C personnel will be briefed on the event and lessons learned including the ineffective use of human performance tools that contributed to this event. Briefing of applicable personnel is scheduled to be complete July 30, 2009.
- The AOV overhaul procedure 0-VLV-416-AOV will be revised to add the following steps: wrench tighten jam lock nuts, verify installation of a lock washer at both jam nuts during positioner reinstallation, wrench tighten connection of linkage to the yoke and positioner, wrench tighten the connection of the turnbuckle to the yoke, ensure Loctite is applied to the threads or a lock washer is installed to prevent the bolt from turning out, verify minimum thread engagement of one diameter of the rod into the turnbuckle if the linkage was disassembled or replaced during maintenance, In addition, a diagram of the proper configuration of the linkage will be added to the procedure. Revision of the procedure is scheduled to be completed by August 31, 2009.
- AOV diagnostic procedure 0-VLV-404-AOV will be revised to add the following steps: wrench tighten jam nuts after diagnostic testing is completed and prior to procedure completion, verify minimum thread engagement of one diameter of the rod into the turnbuckle if the linkage was disassembled or replaced during the diagnostics. A diagram will be added to the procedure of the proper configuration of the linkage. Revision of the procedure is scheduled to be completed by August 31, 2009.
- An additional EOC review will be performed for AOVs with similar feedback mechanisms as BFD-FCV-437. A list of all Unit 2 and Unit 3 AOVs with Bailey AV style positioner with similar feedback mechanisms will be complied and identified valves ranked based on criticality, risk significance and single point vulnerability and CAs issued to inspect the linkage. Additional EOC action is scheduled to be complete by August 3, 2009.
- A list of other critical valves with mechanical feedback arms other than the type of positioner associated with this event will be developed and the preventive maintenance procedures associated with them will be reviewed to determine if there are any latent procedure weaknesses. Any necessary corrective actions (CA) will be assigned to revise procedures as necessary. Review and CA initiation is scheduled to be complete by September 10, 2009.

Event Analysis

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System (RPS) including RT and AFWS actuation. This event meets the reporting criteria because a manual RT was initiated at 01:53 hours, on May 15, 2009, and the AFWS actuated as a result of the RT. The RT did not result in the loss of any safety function. Therefore, there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v). On May 15, 2009, at 03:58 hours, a 4-hour non-emergency notification was made to the NRC for an actuation of the reactor protection system while critical and included an 8-hour notification under 10CFR50.72(b)(3)(iv)(A) for a valid actuation of the AFW System (Event Log # 45069).

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Past Similar Events

A review was performed of the past three years of Licensee Event Reports (LERs) for unit 3 events that involved a RT from a FW FCV failure. No applicable LERs were identified for unit 3. Unit 2 LERs for the previous three years were reviewed and there were four LERs that reported RTs due to MBFP problems: 1) LER-2007-004 reported a RT as a result of low SG levels caused by a loss of MBFPs due to a suction pressure transmitter power supply failure. The cause was an inadequate program to address power supply degradations, 2) LER-2008-001 reported a manual RT as a result of the rapid reduction of MBFP speed control due to radio frequency interference from camera use near the MBFP speed controller processor. A contributing cause was an inadequate procedure for camera use, 3) LER-2008-003 reported a manual RT due to decreasing SG levels and turbine runback. The cause was weak procedural guidance for positioning the arm/defeat switch and failure to follow startup procedures, 4) LER-2009-002 reported a manual RT due to decreasing SG levels caused by the loss of a MBFP. The MBFP was lost due to a failed Autostop oil tube caused by improper installation as a result of poor worker practices. A modification in 1986 failed to provide adequate routing and installation details. Current procedures and engineering standards have been revised and determined to provide adequate guidance. Several of these LERs reported the cause or contributing cause as inadequate procedures but none of them concerned procedures for maintenance of MBFP and FRVs. Therefore, their corrective actions would not have prevented this event.

Safety Significance

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. In response to the event, main FW isolated and the Auxiliary Feedwater System {BA} automatically started. The AFWS actuation was an expected reaction as a result of low SG water level due to SG void fraction (shrink), which occurs after a RT and main steam back pressure as a result of the rapid reduction of steam flow due to turbine control valve closure.

There were no significant potential safety consequences of this event under reasonable and credible alternative conditions. A RT and the increase in SG level is a condition for which the plant is analyzed. This event was bounded by the analyzed event described in FSAR Section 14.1.10, "Excessive Heat Removal Due to Feedwater System Malfunctions." Excessive FW additions is an analyzed event postulated to occur from a malfunction of the FW control system or an operator error which results in the opening of a FW control valve. The analysis assumes one FW valve opens fully resulting in the excessive FW flow to one SG. For the FW system malfunction at full power, the FW flow resulting from a fully open control valve is terminated by the SG high level signal that closes all FW control valves and trips the MBFPs. The SG high water level signal also produces a signal to trip the main turbine. A TT initiates a RT. The analysis for all cases of the excessive FW addition initiated at full power conditions with and without automatic rod control, show that the minimum DNBR remains above the applicable safety analysis DNBR limit, the primary and secondary side maximum pressures are less than 110% of the design values, and all applicable Condition II acceptance criteria are met. For this event, rod control was in automatic and all rods inserted upon initiation of the RT. The AFWS actuated and provided required FW flow to the SGs. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.